November 4, 1991

Docket No. 50-482

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Mr. Bart D. Withers
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

Dear Mr. Withers:

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M80714)

The Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications in response to your application dated June 21, 1991, as supplemented by letter dated September 11, 1991.

The amendment revises Technical Specification 4.6.2.3, "Containment Cooling System," and affected Technical Specification Bases to reduce the minimum required cooling water flow to the containment cooling units during accident conditions. The amendment also changes a monthly surveillance from a verification of flow to the coolers to a verification of valve alignment in order to ensure the coolers will perform their safety function.

A copy of our related Safety Evaluation is enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original Signed By

William D. Reckley, Project Manager
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 50 to NPF-42
2. Safety Evaluation

cc w/enclosures:
See next page

TECHNICALLY CONSISTENT WITH THE LICENSING BASIS

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 4, 1991

Docket No. 50-482

Mr. Bart D. Withers
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

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William D Reckley

William D. Reckley, Project Manager
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
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2. Safety Evaluation

cc w/enclosures:
See next page

cc w/enclosures:
Jay Silberg, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW
Washington, D.C. 20037

Mr. Chris R. Rogers, P.E.
Manager, Electric Department
Public Service Commission
P. O. Box 360
Jefferson City, Missouri 65102

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 311
Burlington, Kansas 66839

Mr. Robert Elliot, Chief Engineer
Utilities Division
Kansas Corporation Commission
1500 SW Arrowhead Road
Topeka, Kansas 66604-4027

Office of the Governor
State of Kansas
Topeka, Kansas 66612

Attorney General
1st Floor - The Statehouse
Topeka, Kansas 66612

Chairman, Coffey County Commission
Coffey County Courthouse
Burlington, Kansas 66839

Mr. Gerald Allen
Public Health Physicist
Bureau of Air Quality & Radiation Control
Division of Environment
Kansas Department of Health
and Environment
Forbes Field Building 321
Topeka, Kansas 66620

Mr. Gary D. Boyer
Director Plant Operations
Wolf Creek Nuclear Operating Corporation
P. O. Box 411
Burlington, Kansas 66839

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Mr. Harold K. Chernoff
Supervisor Licensing
Wolf Creek Nuclear Operating Corporation
P. O. Box 411
Burlington, Kansas 66839

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. NPF-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Wolf Creek Generating Station (the facility) Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated June 21, 1991, and supplemented by letter dated September 11, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-42 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 50, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

G. F. Dick

for Suzanne C. Black, Director
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 4, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 6-15	3/4 6-15
B 3/4 6-2	B 3/4 6-2
B 3/4 6-4	B 3/4 6-4
	B 3/4 6-5

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Two independent groups of containment cooling fans shall be OPERABLE with two fan systems to each group.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one group of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore the inoperable group of cooling fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore at least one group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required containment cooling fans inoperable and one Containment Spray System inoperable, restore the inoperable Containment Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Starting each non-operating fan group from the control room, and verifying that each fan group operates for at least 15 minutes.

2) Verifying that each valve (manual, power-operated, or automatic) in the cooling water flow path serving the containment coolers that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months by verifying that on a Safety Injection test signal, the fans start in slow speed or, if operating, shift to slow speed and the cooling water flow rate increases to at least 2000 gpm to each cooler group.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times* as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The containment isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

*For valves with excessive leakage, refer to Technical Specification 3.6.1.2.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

For reduced pressure tests, the leakage characteristics yielded by measurements L_{tm} and L_{am} shall establish the maximum allowable test leakage rate L_t of not more than L_a (L_{tm}/L_{am}). In the event L_{tm}/L_{am} is greater than 0.7, L_t shall be specified as equal to L_a $(P_t/P_a)^{1/2}$

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.0 psig, and (2) the containment peak pressure does not exceed the design pressure of 60 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 48.9 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 50.4 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained in accordance with safety analysis requirements for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 50.4 psig in the event of a steam line break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of proposed Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979, and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerance on cracking, the results of the engineering evaluation and the corrective actions taken.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 36-inch containment purge supply and exhaust isolation valves are required to be closed and blank flanged during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed and blank flanged during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that the 36-inch containment valves cannot be inadvertently opened, the valves are blank flanged.

The use of the containment mini-purge lines is restricted to the 18-inch purge supply and exhaust isolation valves since, unlike the 36-inch valves, the 18-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment purging operation. Operation will be limited to 2000 hours during a calendar year. The total time the Containment Purge (vent) System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons, e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, should be used to support the additional time requests. Only safety-related reasons should be used to justify the opening of these isolation valves during MODES 1, 2, 3 and 4, in any calendar year regardless of the allowable hours.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2.b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the

SPRAY ADDITIVE SYSTEM (Continued)

solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. The educator flow test of 52 gpm with RWST water is equivalent to 40 gpm NaOH solution. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions. The required design cooling water flow to the Containment Cooling System is verified by the surveillance testing requirements of Specification 4.6.2.3(b) which is performed at 18 month intervals. The testing requirements of Specification 4.6.2.3(a), performed at 31 day intervals, ensure that the fan units and the cooling water flow paths (supply and return) from the Essential Service Water System headers are OPERABLE.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. Operation of the Emergency Exhaust System with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. These Hydro-

3/4.6.4 COMBUSTIBLE GAS CONTROL (Continued)

gen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978.

Adequate mixing of the containment atmosphere following a LOCA is ensured by natural circulation without reliance on a hydrogen mixing systems. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated June 21, 1991, and supplemented by letter dated September 11, 1991, Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station. The proposed changes would revise Technical Specification 4.6.2.3, "Containment Cooling System," and affected Technical Specification Bases to reduce the minimum required cooling water flow to the containment cooling units during accident conditions. The amendment also changes a monthly surveillance from a verification of flow to the coolers to a verification of valve alignment in order to ensure the coolers will perform their safety function. The September 11, 1991, letter provided clarifying information and an additional surveillance requirement to the proposed Technical Specification that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The Service Water System (SWS) is a non-safety related system which provides cooling for plant auxiliaries during normal operation and normal plant shutdowns. The system also supplies cooling water to the safety-related Essential Service Water System (ESWS) during normal operation. During certain conditions, the SWS is isolated and ESWS components are actuated to supply cooling to plant equipment required for the safe shutdown of the reactor or mitigation of design basis accidents. Recent modifications to the SWS and ESWS to address erosion and corrosion concerns have resulted in a reduction in the margins between available cooling water flow rates and the design cooling requirements for various components. To provide sufficient flow rate margins to accommodate future flow balancing to ensure adequate cooling is available for all components, the licensee proposed a reduction in the Technical Specification (TS) required cooling flow rate to the Containment Cooling System (CSS).

The proposed change to TS 4.6.2.3 reduces the required cooling water flow rate to the CCS from the ESWS in its post-accident alignment from 4000 to 2000 gallons per minute (gpm) for each containment cooler group. The proposed flow rate reduction was reanalyzed using plant specific data and contemporary computer codes. The reanalysis demonstrated the conservatisms in the original safety analysis and supported the adequacy of the CCS with the reduced ESWS flow rates.

The evaluation of the reduction in the minimum allowable ESWS flow to the CCS included analysis of the containment pressure and temperature response following a postulated loss-of-coolant accident (LOCA) or main steamline break (MSLB). The revised analysis was performed by the licensee utilizing the CONTEMPT-LT/28 computer code. A comparison of CONTEMPT-LT/28 results to the current containment analyses which were performed using the COPATTA computer code shows very close agreement for those cases using similar input. The licensee's use of CONTEMPT-LT/28 is considered to be acceptable to evaluate the impact of the reduced ESWS flow rate associated with this proposed TS amendment.

The evaluation of a spectrum of LOCA pipe break sizes concludes that the predicted peak containment pressure and temperature were not increased by the reduced ESWS flow rate to the CCS. The lack of impact on the predicted peak containment pressure is due to the limited contribution of the fan coolers in the initial phases of the LOCA. The peak pressure is dependent upon factors such as reactor coolant inventory and energy, containment volume, passive containment heat sinks, and the combined heat removal of containment spray and the fan coolers. The peak containment temperature during a LOCA is dependent upon the same parameters but the contribution of the containment spray in limiting the maximum temperature and introducing a significant temperature decrease upon actuation is more significant than in the pressure determination. The reduced ESWS flow to the fan coolers does delay the long term pressure and temperature reductions. However, as in the original analysis, the containment pressure is reduced significantly below 50 percent of design pressure within the 24 hours following the LOCA. The LOCA containment temperature response remains bounded by the MSLB analyses and the predicted peak pressure remains conservative with respect to the TS 3.6.1.2 P_a value of 48 psig and well below the design pressure of 60 psig.

The analysis of the peak pressure for the MSLB accident predicted a small increase in the peak pressure. The existing analysis limiting MSLB case is a 0.66 square feet split rupture at 25 percent power with a peak pressure of 48.1 psig. The analysis assuming reduced ESWS flow to the fan coolers resulted in a peak pressure of 48.9 psig for the limiting case of a 0.8 square feet split rupture at 50 percent power. In all cases the peak pressure occurs at 1800 seconds after the MSLB which corresponds to the analysis assumption for the isolation of auxiliary feedwater to the faulted steam generator. The MSLB peak pressure analysis demonstrates that the pressure will remain significantly below the design pressure of 60 psig.

The MSLB peak containment temperature in the existing analysis and the reduced ESWS flow analysis occurs immediately prior to the introduction of containment spray into the containment atmosphere. The reduced ESWS flow analysis resulted in a predicted peak temperature of 386.5°F for a double-ended rupture at 50 percent power. This is a slight increase from the previously calculated peak of 384.9°F for a 0.84 square feet break at 75 percent power. This change is not considered to be significant and is more likely caused by analytical differences than by the assumption of reduced ESWS flow to the fan coolers. The licensee reviewed the environmental qualification documentation for equipment located inside containment and concluded that the equipment remains fully qualified for the revised containment environmental conditions associated with the analysis of reduced ESWS flow to the fan coolers.

In addition to the proposed reduction in required ESWS flow to the fan coolers, the licensee has proposed a change to the monthly surveillance which involves a verification of flow rates to the CCS during normal operation (i.e., with the SWS providing flow to the ESWS). The proposed TS would replace the monthly flow verification with a monthly valve alignment verification. The proposed revision is based upon the limited benefit of flow verifications during the normal system alignments and the existence of other surveillances and programs which adequately ensure the availability and performance of the CCS. The capability of the fan coolers to perform their function is considered to be adequately verified by the ESWS flow verification of TS 4.6.2.3.b which is performed at least once per 18 months, the performance monitoring of the coolers which is part of the Generic Letter 89-13 program, and the monthly operation of the fans and valve alignment verification of the proposed TS 4.6.2.3.a. Normal operation of the coolers and monitoring of containment air temperature and cooler leakage also contribute to the verification of CCS performance and integrity.

Based upon its review of the proposed reduction in the minimum required cooling water flow rate to the containment coolers, the staff finds that the change does not significantly impact the containment pressure and temperature response and that relevant design limits continue to be satisfied. For this reason, the staff has determined that the proposed change is acceptable. The proposed revision to the monthly surveillance is also considered acceptable based upon the limited benefit of a flow verification in the non-safety system alignment and the existence of various surveillances and programs which actually ensure containment cooling capability.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding

(56 FR 37596). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: W. Reckley

Date: November 4, 1991